NRC FORM 366 (6-1998) U.S. NUCLEAR REGULATORY COMMISSION

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LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

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TITLE (4)

Personnel Error Causes Two Channels to be in Tripped Condition, Resulting in Reactor Trip

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
										05000
04	23	1999	1999	-- 007 --	01	07	23	1999	FACILITY NAME	DOCKET NUMBER
										05000

OPERATING MODE (9)	1
POWER LEVEL (10)	035

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

	20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)(B)		50.73(a)(2)(viii)
	20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
	20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
	20.2203(a)(2)(ii)		20.2203(a)(4)	X	50.73(a)(2)(iv)		OTHER
	20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)	Specify in Abstract below or in NRC Form 366A	
	20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
John T. St. Martin - Technical Assistant	(716) 771-3641

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE).	NO X				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On April 23, 1999, at approximately 0748 EDST, with the plant in Mode 1 at approximately 35% reactor power and a power escalation in progress after the 1999 refueling outage, Instrument and Control technicians inadvertently pulled fuses from the wrong nuclear instrument channel, causing a reactor trip due to high range flux trip.

The Control Room operators performed the appropriate actions of procedures E-0 and ES-0.1. Following the reactor trip, all systems operated as designed, and the reactor was stabilized in Mode 3.

The underlying cause of the reactor trip was a personnel error, in that fuses were pulled on the wrong nuclear instrument channel. Further evaluation of the event has resulted in more accurately categorizing the root causes.

Immediate corrective action was taken to stabilize the plant in Mode 3. Corrective actions to prevent recurrence are outlined in Section V.B.

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I. PRE-EVENT PLANT CONDITIONS:

On April 23, 1999, the plant was in Mode 1 at approximately 35% reactor power. A power escalation was in progress as the plant started up from the 1999 refueling outage. Instrument and Control (I&C) technicians were in the process of adjusting nuclear instrument system (NIS) power range (PR) trip setpoints for the four (4) PR channels, as required during initial power escalation after a refueling outage. The I&C technicians had completed adjustments for NIS PR channel N-41 at approximately 0650 EDST, and had defeated channel N-42 at approximately 0700 EDST. Channel defeat involves placing numerous bistables into the tripped condition in protection racks in the Control Room, and results in a similar number of Main Control Board (MCB) annunciator alarms. The I&C technicians then proceeded to the NIS racks (in a different location in the Control Room) to continue with trip setpoint adjustment.

II. DESCRIPTION OF EVENT:

A. DATES AND APPROXIMATE TIMES OF MAJOR OCCURRENCES:

- o April 23, 1999, 0529 EDST: Setpoint adjustment is started for NIS PR channel N-41.
- o April 23, 1999, 0650 EDST: Setpoint adjustment is completed for NIS PR channel N-41.
- o April 23, 1999, 0700 EDST: Setpoint adjustment is started for NIS PR channel N-42.
- o April 23, 1999, 0748 EDST: Event date and time.
- o April 23, 1999, 0748 EDST: Discovery date and time.
- o April 23, 1999, 0749 EDST: Control Room operators verify both reactor trip breakers open and verify all control and shutdown rods inserted.
- o April 23, 1999, 0754 EDST: Control Room operators manually stop the operating main feedwater pump to limit a reactor coolant system cooldown.
- o April 23, 1999, 0756 EDST: Control Room operators manually close both main steam isolation valves to limit a reactor coolant system cooldown.
- o April 23, 1999, 0841 EDST: Plant is stabilized in Mode 3.

B. EVENT:

On April 23, 1999, the plant was in Mode 1 at approximately 35% reactor power. A power escalation was in progress as the plant started up from the 1999 refueling outage. Instrument and Control (I&C) technicians were in the process of adjusting nuclear instrument system (NIS) power range (PR) trip setpoints for the four (4) PR channels, as required during initial power escalation after a refueling outage. The I&C technicians had completed adjustments for NIS PR channel N-41 at approximately 0650 EDST, and had defeated channel N-42 at approximately 0700 EDST. Channel defeat involves placing numerous bistables into the tripped condition in protection racks in the Control Room, and results in a similar number of Main Control Board (MCB) annunciator alarms. The I&C technicians then proceeded to the NIS racks (in a different location in the Control Room) to continue with trip setpoint adjustment.

At approximately 0748 EDST, the I&C technicians prepared to pull the fuses for NIS PR channel N-42 at the N-42 drawer. They approached a drawer (for channel N-43, by mistake) and pulled the fuses. Pulling these fuses tripped the high flux trip bistables for channel N-43. With the channel N-42 bistables already previously tripped, the reactor tripped on 2/4 NIS PR high flux range trip. In addition to the MCB annunciators already in alarm from the N-42 channel defeat, the Control Room operators acknowledged MCB annunciators D-2 (Power Range High Range Reactor Trip 2/4 108%) and D-10 (Power Range Lo Range Reactor Trip 2/4 24%), indicating a reactor trip from NIS PR channels.

The Control Room operators performed the immediate actions of Emergency Operating Procedure E-0, "Reactor Trip or Safety Injection". They transitioned to Emergency Operating Procedure ES-0.1, "Reactor Trip Response", when it was verified that both reactor trip breakers were open, all control and shutdown rods were inserted, and safety injection was not actuated or required.

During the performance of ES-0.1, the Control Room operators noted that a reactor coolant system (RCS) cooldown was occurring, due to addition of feedwater and a low decay heat level after the recently completed refueling outage. Due to this RCS cooldown, the Control Room operators manually stopped the operating main feedwater (MFW) pump at approximately 0754 EDST (after manually starting the motor-driven auxiliary feedwater (AFW) pumps) and closed both main steam isolation valves (MSIVs) at approximately 0756 EDST. These actions mitigated the RCS cooldown.

The plant was stabilized in Mode 3 at approximately 0841 EST and the Control Room operators transitioned to normal plant operating procedures.

C. INOPERABLE STRUCTURES, COMPONENTS, OR SYSTEMS THAT CONTRIBUTED TO THE EVENT:

None

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TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

D. OTHER SYSTEMS OR SECONDARY FUNCTIONS AFFECTED:

None

E. METHOD OF DISCOVERY:

This event was immediately apparent due to Main Control Board indication of the reactor trip, due to plant response and alarms and indications in the Control Room.

F. OPERATOR ACTION:

After the reactor trip, the Control Room operators performed the appropriate actions of Emergency Operating Procedures E-0 and ES-0.1. Both motor-driven auxiliary feedwater pumps were manually started. The MFW pump was stopped and the MSIVs were manually closed to limit further RCS cooldown. The plant was stabilized in Mode 3.

Subsequently, the Control Room operators notified higher supervision and the NRC per 10 CFR 50.72 (b) (2) (ii), non-emergency four hour notification, at approximately 1016 EDST on April 23, 1999.

G. SAFETY SYSTEM RESPONSES:

All safeguards equipment functioned properly.

III. CAUSE OF EVENT:

A. IMMEDIATE CAUSE:

The immediate cause of the reactor trip was achieving the 2/4 reactor protection system (RPS) trip logic for NIS PR high flux range trip.

B. INTERMEDIATE CAUSE:

The intermediate cause of achieving 2/4 RPS trip logic was the bistable for NIS PR channel N-43 high flux range being tripped due to pulling the fuses for channel N-43, with the channel N-42 bistable already tripped for setpoint adjustment.

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C. ROOT CAUSE:

Further evaluation of this event has been conducted. The evaluation concluded that the underlying cause of the pulling of fuses for channel N-43 was a failure to self-check and peer-check during performance of maintenance.

As a result of this failure, I&C technicians initiated activities on the wrong power range channel and caused a reactor trip. Causal factors identified included:

- o Written Communication, in that independent verification was not required for the particular steps in the calibration procedure
- o Interface Design, in that the four power range drawers are adjacent to each other with identical configurations
- o Work Practices, in that STAR (stop, think, act, review) and self-verification were used inadequately
- o Verbal Communications, in that the pre-job brief and three-way communications between the two I&C technicians who were performing work on channel N-42 were inadequate

This error was a cognitive error on the part of two I&C technicians, who intended to pull the fuses on the drawer for NIS PR channel N-42. They approached the second drawer from the west (drawer for channel N-43), rather than the second drawer from the east (drawer for channel N-42), and did not recognize that they were on the wrong NIS PR drawer. This error was inadvertent, and was contrary to the approved procedure. There are no unusual characteristics of the work location (Control Room).

IV. ANALYSIS OF EVENT:

This event is reportable in accordance with 10 CFR 50.73, Licensee Event Report System, item (a) (2) (iv), which requires a report of, "Any event or condition that resulted in a manual or automatic actuation of any engineered safety feature (ESF), including the reactor protection system (RPS)". The reactor trip was an actuation of the RPS and AFW pump starts are actuations of an ESF component.

An assessment was performed considering both the safety consequences and implications of this event with the following results and conclusions:

There were no operational or safety consequences or implications attributed to the reactor trip because:

- o The two reactor trip breakers opened as required.

- o All control and shutdown rods inserted as designed.
- o The plant was stabilized in Mode 3.
- o The Ginna Station Improved Technical Specifications (ITS) Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) were reviewed with respect to the post trip review data. The following are the results of that review:
  - a. Pressurizer (PRZR) pressure decreased below 2205 PSIG during the transient after the reactor trip. During this time a thermal power step >10% occurred due to the reactor trip, which is within the limits of ITS LCO 3.4.1. Therefore, compliance with ITS was maintained. The RCS temperature DNB limit (577.5 degrees F) was not approached. Additional mitigation was provided by stopping the MFW pump and closing the MSIVs. Minimum PRZR pressure was approximately 2184 PSIG, and PRZR pressure was restored > 2205 PSIG within five minutes.
  - b. After the reactor trip, the RCS cooled down to approximately 541 degrees F and was subsequently stabilized at 547 degrees F. The cooldown was within the limits of ITS LCO 3.4.3. In addition, the required shutdown margin was maintained at all times during the RCS cooldown.

Based on the above and a review of post trip data and past plant transients, it can be concluded that the plant operated as designed, that there were no unreviewed safety questions, and that the public's health and safety was assured at all times.

V. CORRECTIVE ACTION:

A. ACTION TAKEN TO RETURN AFFECTED SYSTEMS TO PRE-EVENT NORMAL STATUS:

A different crew of I&C technicians was assigned to perform trip setpoint adjustments for NIS PR channels N-42, N-43, and N-44, and the adjustments were completed.

B. ACTION TAKEN OR PLANNED TO PREVENT RECURRENCE:

- o The two I&C technicians involved in this event were temporarily relieved of these calibration responsibilities to participate in the event investigation, and were counseled concerning this event.
- o To address the failure to self or peer-check, a Nuclear Training Work Request (NTWR 99-0504) has been initiated to conduct STAR training for I&C technicians

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- o The importance of self and peer-check was discussed at an I&C shop meeting.
- o To address Written Communication, procedures for calibration of NIS bistables and axial flux monitoring will be revised to add additional independent verification steps.
- o To address Interface Design, labeling of the NIS cabinets has been improved to include the use of protection channel colors to identify separate cabinets.
- o The standard for I&C pre-job briefs has been upgraded to include the use of red safety barrier requirements.
- o To address Verbal Communications and Work Practices, toolbox training will be conducted for I&C technicians to reinforce management expectations on work practices and verbal communications.
- o Should additional corrective actions be identified, a supplement to this LER will be issued.

VI. ADDITIONAL INFORMATION:

A. FAILED COMPONENTS:

None

B. PREVIOUS LERs ON SIMILAR EVENTS:

A similar LER event historical search was conducted with the following results: LERs 90-012 and 93-007 were similar events (reactor trip) with a similar root cause (personnel error directly caused a reactor trip).

C. SPECIAL COMMENTS:

None